6/24/80

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	§	
	§	
HOUSTON LIGHTING & POWER COMPANY	§	Docket No. 50-466
	§	
	§	
(Allens Creek Nuclear Generating Station, Unit No. 1)	§	
	§	
	§	

APPLICANT'S RESPONSE TO JOHN F. DOHERTY'S FOURTEENTH SET OF INTERROGATORIES TO HOUSTON LIGHTING & POWER COMPANY

In response to the interrogatories propounded by John F. Doherty, Houston Lighting & Power Company (Applicant) answers as follows:

Contention 11 - Spent Fuel Pool

1. What is the highest anticipated enrichment/of fuel for which the fuel pool is designed?

Response: Spend fuel storage racks are designed for fuel with an average enrichment of 3.15 weight percentage with U235.

2. Will any of the structural material be neutron-absorbing in the spent fuel pool? If so, what material, where in the racks and how long will the material have the capacity to absorb neutrons.

Response: The rack structure material, which is stainless steel 304, will absorb neutrons and credit is

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taken for this effect in the critically analysis. The stainless steel structure has the capacity to absorb neutrons throughout the 40 years licensed life time of the rack.

3. Will applicant's spent fuel pool be in compliance with ANSI N210-1976? If any non-compliance, please note.

Response: The relationship between the design of the SFPCCS and ANSI N210-76 is detailed in PSAR Section 9.1.3.

4. Referring to "2" above, will any credit be taken for neutron absorbing material in the Keff calculations for the spent fuel pool? If so, how much?

Response: The criticality calculation considers fuel assembly to be at the most reactive point in the lifetime, with no credit for U235 burnup or fission product neutron absorbtion. No credit for any neutron absorbing material is considered in the spent fuel pool or in the racks of the Keff calculation for the spent fuel pool, except for the stainless steel of the rack structure as noted in the response to 2 above. For the ACNGS spent fuel pool racks, the stainless steel subtracts approximately .125 from K per .1 inch.

5. A concern expressed in NUREG-0626, is that the RHR is used for both shutdown cooling and fuel pool cooling, and hence, "Fuel pool cooling systems should be self-sufficient" (P. 1-15, NUREG-0626, "Generic Review of Boiling Water Reactors in view of the TMI-2 Lessons Learned", paraphase title).

A. Has applicant committed to such a self-sufficient system?

B. Has applicant done, or is applicant aware of any risk assessment for the use of the RHR for fuel pool cooling, and if so, please make available to me.

C. If the RHR is required for prolonged duty following either a design based or other accident what system will be available to take over RHR pump functions? What is the technical specification of that system with regard to length of service without ceasing to operate for any reason?

Response: A. The Fuel Pool Cooling & Cleanup System is self-sufficient as it is currently designed. The use of the RHR system in conjunction with the FPCCS is limited to certain abnormal conditions, e.g., in the event that the entire core must be unloaded from the RPV and stored in the spent fuel pool.

B. No, no such assessment has been done beyond that which appears in Sections 5.5.7 (RHR System) and 9.1.3 of the ACNGS PSAR. However, the postulated conditions under which the RHR might conceivably be used to augment the FPCCS heat removal capability are mutually exclusive with those which would require its use as a safety system.

C. The RHR System is not required in the FPCCS assist mode during post-accident conditions. There are two complete trains of RHR pump/heat exchangers available. See p.9.1-7 of the ACNGS PSAR. Technical specifications for ACNGS have not yet been developed.

6. Is removal of spent fuel in event the crew of the plant must shut-down and leave a possibility in emergency plans?

Response: No. There is no event which would require complete evacuation of the site. The control room is habitable during all accidents and normal operation of the spent fuel pool can be maintained from the control room.

7. Are there any emergency plant for spent fuel cooling if the plant crew must leave the site?

Response: No. See 6 above.

8. Are the Spent Fuel Pool and Fuel Handling Pool equally coolable? (That is, is each of the same capacity to deal with the decay heat of the spent fuel rods?)

Response: Yes, the water in both pools is in free communication by virtue of the partial height weir wall between the pools, by the skimmers and scruppers, and by the diffusers which return the cool water to the pools.

9. On page 9.1-9 of the PSAR, it says, "No inlets, outlets, or drains will be provided that might permit the pool to be drained below a safe shielding level. Lines extending below this level will be equipped with redundant siphon breakers, check valves or ...". Are there any lines in the viscinity [sic] of the SFP which would permit this? If so, please indicate where so that they can be placed on drawings 95-1C-20 of the PSAR, Amendment 40. Section A-A, and C-C.

Response: The commitment will be honored as stated in the PSAR. All lines will enter and leave above this water level and suitable siphon breakers will be provided. The lines are shown in PSAR figure 9.1-3a.

10. Does Applicant agree with Rg. Guide 1.12, part B.1 "... loss of water from a fuel storage pool could cause overheating of the spent fuel and resultant damage to fuel cladding integrity and could result in release of radioactive materials to the environment"? If not, please give reason or reasons for not agreeing.

Response: Applicant's position on Regulatory Guide 1.13 is given in ACNGS PSAR Appendix C. Due to the possibility that the fuel could overheat as stated in the cited Regulatory Guide, ACNGS fuel pool is designed in accordance with the response to Interrogatory #9 above to prevent just such an occurrence.

11. Will any of the SFP alarm systems be designed to alarm to an off-site location directly, such that an emergency crew could come to the scene and attempt to correct the malfunction of the SFP protection systems in the event a previous accident has forced evacuation of the site? (See Board Order, Page 16)

Response: No. The FPCCS is designed to run without constant operator attention. FPCCS performance is monitored locally and in the Control Room. Control of the FPCCS is done from the Control Room.

12. What is a fuel specialty rack? See: P. 3.2-28, (Am. No. 53). Table 3.2-1, PSAR.

Response: Fuel Specialty Racks are those racks provided in the FHB Fuel Handling Pool for storing or handling fuel or pieces of equipment of a more specialized nature. They are described in the ACNGS PSAR Section 9.1.4.1.7.

13. Does Applicant accept or reject the statement: water reacts chemically with heated zirconium to produce (1) heat, and (2) possible explosions.

Response: Applicant neither accepts nor rejects the statement. Exothermic chemical reactions between steam and very high temperature zirconium (not found in a normal operating reactor) can occur producing hydrogen gas which, if allowed to mix with oxygen in the presence of an ignition source, could burn or explode.

14. Does applicant accept the statement that the heating of zirconium in 13 (1) above may produce sufficient heat to raise the temperature of the fuel to above its melting temperature, 2350°F?

Response: Applicant neither accepts nor rejects the statement. Interrogatory 13 does not adequately define the circumstances of the postulated event rendering a judgment as to the maximum temperature that could be obtained impossible.

15. Does Applicant take the position its SFP is single failure criterion adequate.

Response: Yes. The decay heat removal capability for the spent fuel pool is single failure proof.

16. To what temperature is the spent fuel building ventilation system able to keep operating, according the specifications for suppliers of such equipment? That is, how hot can the building get before the system will breakdown and cease functioning?

Response: See PSAR Section 9.4.5, fig. 9.4-4 and table 3.11-3.

17. Is Applicant's SFP rack considered one of "high density" by the NRC?

Response: Yes.

18. Does Applicant plan to remove the fuel channels that surround each BWR fuel assembly prior to storage in the SFP?

Response: In general, the answer is "yes" for long term storage. There may be short periods wherein the spent fuel will await dechanneling in the fuel handling pool.

<u>Contention 12</u> - Rod Control and Information System, RCIS

1. Referring to 7.7-9 of the PSAR, how are bypass conditions logged in the process computer?

Response: Bypass conditions are logged in the history file (magnetic tape) and printed out by the line printer.

2. When bypass conditions exist what is the alarm? Bell, flashing light?

Response: Bypassed conditions can be alarmed on the CRT.

3. Describe the non-indicating flow [sic] switch installed on the scram discharge volume which blocks rod removal on high water level.

Response: Non-indicating float switches will not be used at Allens Creek. Discharge volume water level will be monitored by a level transmitter which measures water level by sensing the water level pressure in the discharge volume. Water level is monitored throughout the anticipated range (low to high) and appropriate automatic action is taken when the water level reaches the trip unit set point. The PSAR at this time does not reflect this change but will be updated in the near future.

4. In Applicant's RCIS will the rod withdrawal operations be stopped if the position of a single rod is unknown?

Response: Yes.

5. In your reply to my Interrogatory number 4 of set #4, you indicated you expected some BWR/6's to be operating before ACHGS such that you thought you would gain experience. Is the <u>Grand Gulf</u> reactor identical to the planned ACNGS with regard to this system?

Response: The nuclear steam supply systems are the same except for some size related differences. Grand Gulf is a larger core and has more CRD's.

6. How many of the following are in the ACNGS core? a. SRM channels, b. IRM channels, c. APRM channels.

Response: a. SRM channels: see PSAR sect. 7.6.1.6.3.1.1

b. IRM channels: see PSAR sect. 7.6.1.6.4.1.1

c. APRM channels: see PSAR sect. 7.6.1.6.6.1.1

Contention 25 (First Part) See Board Order, 3/11/80 p. 20-21.

1. What is the size of the inlet flow hole at the bottom of the fuel assembly?

Response: There are three sizes of orifices: 1.448" or 1.257" diameter for the peripheral bundles and 2.43" for the other bundles.

2. Are there debris screens in the core and fuel assemblies to catch falling bits of material from any events, such as in PWRs. If so, describe their mesh, and their location above the bottom of the fuel rod.

Response: No.

Contention 27 - Pedestal Concrete

1. What is the thickness of steel that is part of the pedestal and designated: "Steel RPV Pedestal with Concrete Fill for Steel Structure" (See Figure 3.8.5) in Fig. 3.8-3 of the PSAR?

Response: The thickness of the inner and outer shells of the RPV pedestal are:

3 inches between elevations 146.42' and 132.00'

1.5 inches between elevations 132.00' and 115.00'

2. What are the physical properties of the concrete in the RPV pedestal

a. Density?

b. Melting temperature of the solid phase?

c. Heat of Fusion?

Response: The physical properties of the concrete in the RPV pedestal are:

a. Density = 140 pcf

b. Not determined.

c. Not determined.

3. Will any concrete in the pedestal be poured using aluminum pipes or aluminum ways?

Response: No.

4. In section 3.8.3.1.7 PSAR applicant states it "may" fill the annulus between the steel cylinders with concrete. Has applicant made this determination. If so, what is the name or designation of the chosen material?

Response: Yes. The annulus between the steel cylinders will be filled with non-reinforced concrete. The material under consideration is ordinary concrete or grout.

<u>Contention 15</u> - WIGLE Code

1. What was the outcome of NRC review of the General Electric Program to analyze the Control rod drop accident and revise NEDO 10,527 into a three dimensional analysis?

Response: NEDO-10527 has been reviewed and approved by NRC. Applicant is unfamiliar with any attempt to revise the rod drop analysis into a three dimensional analysis.

4. According to R. L. Crowther of General Electric (Transactions of the American Nuclear Society 32, June 3, 1979, P.724) thermal hydraulic representations in steady-state and transient conditions are the most difficult feed-back to represent in the BWR core simulator.

a. What causes the most difficulty?

b. What areas do you believe represent the most danger to the reactor system if incorrectly represented?

c. What steps are being taken toward decreasing this difficulty?

Response:

a. This feedback is difficult to represent because of the large number of variables and the large amount of data required.

b. None of the representations is so difficult that it bears on the conservatism in safety analyses.

c. GE compares operating reactor data with analytical models and results on a continuous basis.

5. Can delayed neutron contribution be neglected in WIGLE reactor excursion calculations?

Response: See response to 3 above.

Contention 38 (b) - Cold Shutdown in 24 hrs.

1. Does Applicant take the position that following a design based LOCA it can achieve cold shutdown in 24 hours without current plans being changed?

Response: In accordance with 10 CFR 50.46, ECCS must be operable and keep the core cool and covered. "Cold shutdown in 24 hours" is not applicable for a post LOCA situation.

2. Can Applicant achieve cold shutdown in 24 hours using a single residual heat removal (RHR) system pump?

Response: Yes.

3. Are other pumps available to remove decay heat from the core in the event of a design based LOCA if, (a) One RMR pump out of service? (b) two RHR pumps out of service?

Response: See response to 1 above.

4. In Amendment #72 to Hatch Unit 1, an operating BWR, Georgia Power requested a 9% reduction in the Tech Specs in RHR pump head pressure due to pump wear.

a. What is Applicant's current system head for each RHR pump?

b. By how much is this estimated to decrease due to pump wear over (a) 10 years? (b) 20 years? (c) 30 years? (d) 40 years?

c. Are any provisions available for installation of and [sic] additional RHR pump in case any criterion for bringing reactor to cold shutdown cannot be met in the future?

Response:

a. See PSAR figure 5.5-10.

b. Pump degradation due to wear is a function of running conditions, running time and the maintenance program. Accordingly it is not possible to estimate pump wear over the periods specified.

c. It is, of course, possible to add an additional pump; there are no present reasons or plans for doing so.

5. Has General Electric received a definition of "cold shutdown" satisfactory to it for use in Reg. Guide 1.139, Reg. Position C. 1, or does it still request a revision as per a letter of 7/27/78 from G.G. Sherwood to Docketing and Service (NRC)?

Response: The GE design is based upon the definition of cold shutdown contained in NUREG-0123, "Standard Technical Specifications for GE BWR's."

6. What is Applicant's definition of "cold shutdown"?

Response: See response to 5 above.

7. Has HL&P decided to enter, or initiate or enter any rulemaking proceedings on the post-TMI suggestion to have a 24 hour limit on time to reach cold shutdown as it indicated in its response to H. Denton's (NRC) letter by E. A. Turner (HL&P) of 8/19/79 (sent to all parties) in regard to NUREG-0578 P. A-63? (The letter is designated AC-HL-AE330 for retrieval purposes)

Response: The referenced "post TMI suggestion" deals with proposed revisions to Limiting Conditions for Operation requirements in plant technical specifications. At such time as a rulemaking hearing is convened on this issue, HL&P may or may not participate in the proceeding.

8. How does G.E. propose to define cold shut-down in its comment on Regulatory Guide 1.1.39, "Guidance for RHR"?

Response: See response to 5 above.

<u>Contention 39</u> - Fuel Rod Balooning [sic]

1. To what pressure(s) will ACNGS fuel rods be filled before insertion in the ACNGS core?

Response: The Allens Creek initial core fuel rods will be filled with helium to a pressure of 3 atmospheres before insertion into the core.

2. Has General Electric modified its calculation of fuel blocking by rod swell to meet 10 CFR requirements since the change from 7 X 7 fuel to 8 X 8 fuel? Make available any documents showing this please.

Response: Data and calculations on fuel rod swelling and flow blockage are contained in the document "General Electric Company Analytical Model for Loss-of-coolant Analysis In Accordance With 10CRF50 Appendix K." NEDO-20566 Jan, 1976. This document, which contains data on both 7 X 7 and 8 X 8 fuel, indicates that the maximum cross sectional blockage is less in the 8 X 8 design.

3. After three years and approximately 30,000 MWd/tonne U fuel burn-up, will fuel rod gas pressure exceed 215 psia at 25°C? If no, please state what the expected pressure will be.

Response: At a peak pellet burn-up of approximately 30,000 MWD/MT, the fuel rod internal pressure is calculated to be 110 psia in the 550°F reactor environment. This corresponds to a pressure less than 65 psia at 25°C.

4. Generally, does irradiation of cladding reduce ductility and hence the possibility of strain caused blockage within BWR fuel channels?

Response: It is correct to assume that irradiation increases cladding strength and reduces ductility. However, with the high temperatures postulated to occur under loss-of-coolant accident conditions, the irradiation strength of the cladding would be annealed out and cladding performance (ballooning, etc.) would be similar for both irradiated and unirradiated cladding. See figure I.B.2.4 of NEDO-20566.

5. During a shutdown period proceeding to cold shutdown post-LOCA, how would pressure be relieve [sic] in the RPV? Would operation of the SRV's be expected to be fairly continuous as challenged?

Response: Post LOCA shutdown is described in PSAR Section 6.3 and in the pertinent accident analyses in chapter 15.

6. How much clad swelling <u>can</u> occur without significantly impairing the effectiveness of the ECCS? Cite a publication used to support the response please.

Response: See NEDO-20566.

<u>Contention 24</u> - Rod drop accident

1. How is "hot stand-by defined by General Electric Co. currently, and give the minimum and maximum coolant temperatures for the status.

Response: Hot standby is defined as the condition with the reactor mode switch in startup/hot standby position and the reactor coolant above 212°F.

2. Does ACNGS use:

(a) Axial placed gadolinium poison?

(b) Varying fuel enrichment in each bundle?

(c) Poison curtains?

Response:

(a) Yes.

(b) Yes.

(c) No.

3. Since a stuck, unconnected control rod may vibrate, what is the minimum vibration amplitude of a Control Rod applicant believes an in-core neutron monitor can detect?

Response: In-core neutron monitors detect neutron flux, not vibration amplitudes.

4. Referreing to 15.1.38.5.1.2.1 of the PSAR, does Applicant still take the position, "The peak enthalpy results of the design based control rod drop accident, are less than the 280 cal/gram design limit with all exposures"?

Response: Applicant's position is described in Section 15.1.38.5.1.2.1 of the PSAR.

5. What is the peal fule [sic] enthalpy for a 2.0% control rod drop at 10% power at 5 ft./sec. for a BWR/6? (This can be compared with th 397 cal/gram answer for other cores, shown on P.4 Sec. 2.4 of the Addendum to NEDO 10,527(1)).

Response: See NEDO 10527, Supplement 1, section 2.4, fig. 2-3.

Contention 29 - Ultimate Heat Sink Inadequacies

1. Has Applicant submitted additional design material (plans) for demonstrating the Ultimate Heat Sink will not have failure of man-made structures? (a) If so, what are these plans generally? (b) Are any of these additional plans to include retaining walls?

Response: No. (a) and (b) Not applicable.

2. Has Applicant committed to removal of all clay from the bottom of the excavation for the causeway embankment of the ultimate heat sink as suggested by the Corps of Engineers in SER, Sup. #2, App. E? If not, for what reasons, please?

Response: It is anticipated that clay will be excavated to the depth recommended by the Corps of Engineers, but no commitment with the Staff has yet been made.

Contention 35 - Welders

1. At the South Texas Nuclear Project as noted in Region IV Office of Inspection & Enforcement (NRC) report 50-498/79-08; 50-499/79-08 of 5/15-5/18/79, on page 4 ("b. Observation of Work") it appears there was no monitoring procedure for review of reactor cooling system piping welding operations. Has Applicant determined the cause of this lack? What was it? Were the welders aware such documents should be available?

Response: A response to I & E report 79-08 was filed with Region IV on September 10, 1979, (E.A. Turner to W.C. Sidell). A copy of this response will be made available at the EDC.

<u>Contention 42</u> - Valve position Indication

1. Does Applicant or General Electric have a single way to accomplish the "committed objective" (See Board <u>Order</u> 5/11/79) of this contention as of the date of these interrogatories?

Response: There are available several designs to meet the objective.

2. With a single stuck open safety relief valve and only low pressure Emergency Core Cooling System available, will operator have to actuate the Automatic depressurization system to prevent any uncovering of the core in the ACNGS.

Response: No.

<u>Contention 45</u> - Lateral Support for the Core

1. Does applicant maintain fuel channel boxes will provide lateral support sufficient to prevent lateral core movement from causing the problem alleged?

Response: No.

2. If so, will applicant rely on no other documents buy NUREG/CR-1018 and NEDO 21, 175-P? If "no" list any other docuements [sic]?

Response: The complete list of relevant documents is presently indeterminate.

3. At the issue in this contention is to be resolved on a plant by plant basis, have you submitted any material to the PSAR to cover this accident result? Please indicate where this is in the PSAR and Its amendment number.

Response: See PSAR Section 3.9.1.5. This material was submitted in Amendment 35.

4. In NUREG/CR-1018, P. 13 (D-3(c)), the contractor report states that a lateral LOCA force requires an additional margin of support in the fuel assemblies above that for the Safe Shutdown Earthquake by about 30% This lateral force is due to, "flashing which occurs near the end of the sub-cooled blowdown portion of the LOCA transient.", and the report suggests it should be included in th LOCA analysis.

(a) Does Applicant take the position, or otherwise maintain the ANGS core contains support against vibration greater than 30% more than the Safe Shutdown Earthquake?

(b) Does the above described flashing occur in the ACNGS core following the actuation of the ECCS in response to a LOCA?

(c) If it does, kindly indicate where in the literature the calculation of its force is to be found?

(d) Also indicate where in the literature it is concluded such flashing will be accomodated in the internal core structure design of either the ACNGS or a 238" BWR/6.

Response:

(a) The ACNGS core is designed to accomodate combined SSE and LOCA loadings as described in PSAR Section 3.9.1.5 and is designed to meet ASME III Subsection NG.

(b) Some of the reactor coolant will flash to steam in the event of a large break LOCA. This "flashing" will occur prior to ECCS actuation.

(c) See PSAR Section 3.9.1.5.

(d) See PSAR Section 3.9.1.5.

<u>Contention 46</u> - Xenon transients

1. Does the velocity limiter on a BWR control rod encounter water on the control rod's descent or air when the rod is lowered?

Response: Water.

2. Does the ACNGS control rod drive system have an uncoupling rod for use with each control rod or several control rods? If so, what is the purpose of the uncoupling rod.

Response: Each control rod can be uncoupled. To uncouple the control rod a special tool is used to disengage the control rod from the control rod drive by raising the control rod coupling release handle (figure 4.2-12 of the PSAR; see also figure 4.2-15). The uncoupling tool fits all of the control rods. Rods are uncoupled for replacement and maintenance purposes.

3. If the answer to "2"Above is affirmative indicate where in the literature this Intervenor may find information on how control rods are uncoupled from their drivers. If there is a PSAR citation, please give that to the page number.

Response: See response to 2 above.

4. Do the Source Range Monitors move, or are the SRMs at a fixed height from the bottom of the core?

Response: The SRM's are described in PSAR section 7.6.1.6.3.

5. What effect does pre-conditioning the fuel have on the results of the control rod drop accident calculations?

Response: None.

6. Can the drop of a single notch at any position in core have a rate of 5 feet/second? (Note: NEDO 10,527, (P. 4) gives some figures of 5 feet/sec. drops leading to 397 cal./gm energy insertion in the fuel rods)

Response: No.

7. How frequently will the control rod blades be cnanged in the ACNGS according to technical specifications?

Response: Technical specifications have not been written for ACNGS.

8. (a) Must control rod blades be removed for inspection?

(b) How often must control rods be visually inspected?

Response:

(a) If a control rod has to be visually inspected (at present there is no requirement to do this) it would be removed from the reactor.

(b) Control rod blades are tested, in core, to determine their scram reactivity. Once the reactivity has desceased to non-acceptable limits (established by technical specification requirements) the CRD blades are replaced.

9. Does NEDO 21,231, currently represent the General Electric Position on the most conservative but reasonable method to withdraw control rods, i.e. with second 25% of rods in banked mode? Does Applicant accept the G.E. position?

Response: NEDO 21,231 represents the current GE position with regard to control rod withdrawal sequence. Applicant has no present reason to take exception to thses recommendations.

10. Does Applicant maintain that it is absolutely impossible for peak of fuel rod enthalpy to reach 500 cal/gram?

Response: Applicant maintains that it is virtually impossible to assert that _anything_ is "absolutely impossible."

11. Does Applicant maintian [sic] that a five second period cannot be caused by the fall of a control rod of a single notch distance?

Response: No.

12. Does Applicant maintain that a five second period cannot be caused by the fall of a control rod of two or more notch distances?

Response: No.

13. Does Applicant maintain that a five second priod cannot be caused by the fall of a control rod of four or more notch distances?

Response: No.

14. Does Applicant maintain no fuel damage will occur if a control rod falls a single notch under _any_ core conditions?

Response: No.

15. Does Applicant maintain no fuel damage will occur if a control rod falls two or more notch distances under _any_ core conditions?

Response: No.

16. Does Applicant maintain no fuel damage will occur if a control rod falls four or more notch distances under _any_ core conditions?

Response: No.

17. Are control rod blades removed from their drives when the control rod is fully inserted during refueling?

Response: Yes.

18. Are control rod blades ermoved from their drives for any reason other than replacement? If so, for what reasons.

Response: No.

19. When in the history of operation of the plant are control rods disengaged from their drives?

Response: See response to 8 above.

20. What is the mass and weight of an ANCGS control rod? Include no material below the coupling with the control rod drive.

Response: A typical ACNGS control rod is 173.986 in. maximum length by approximately 10 in. wide. It weighs approximately 218 lbs.

21. What is the maximum speed attained by a control rod on insertion?

Response: Rod speeds may be calculated from PSAR figure 15.1.1-1. See also, PSAR Section 4.2.3.2.2.4.

22. What is the calculated force exerted on the control rod-control rod drive coupling or locking when a rod is inserted from fall out during a SCRAM?

Response: Force on the rod is a function of the differential between vessel pressure and drive pressure and the area the pressure is acting on (approx. 4 in^2). Scram under zero vessel pressure conditions would exert a force of approximately 1750 $lb/in.^2 \times 4\ in.^2$ or 7000 lbs.

23. Referring to "22" above, is there any cable or other linkage other than the locking mechanism to resist or absorb some of this calculated force from SCRAM on the locking mechanism? If so, describe please, and give a PSAR reference.

Response: The drive is slowed down at the top of its stroke both hydraulically and mechanically. The hydraulic slowing down of the drives is described in Section 4.2.3.2.2.4 of the PSAR. The buffer holes that decrease the pressure differential and slow down the rod are shown in PSAR fig. 4.2-18 item 53. Mechanical slowing down of the piston occurs as the drive piston goes all the way in. As shown on figure 4.2-18, as the index tube (item 26) inserts the control rod into the vessel, the flange face where the index tube and the drive piston (item 24) makes contact with the spring washers (item 30) compresses the washers and slows down the drive.

24. Is Applicant planning on using grouped control rods withdrawal sequences with the RC & IS? If so, please describe and give PSAR reference.

Response: Yes. A description can be found in PSAR Section 7.7.1.1.

Contention 48 - Control Rod Drive Return Line Removal

1. In Section 5.4.2.3.7 of the PSAR, "It states, "The recirculation inlet nozzles... and the control rod drive (CRD) hydraulic return line all have thermal sleeves

(a) Are these sleeves of A-508 Stainless steel? If not, of what material are they?

(b) Does Applicant plan a CRD return line or not?

(c) How long have any of the thermal sleeves mentioned in 5.4.2.3.7 of the PSAR been used in the CRD return line of an operating BWR?

Response:

(a) The recirculation inlet nozzle thermal sleeve is made of 31[illegible].

(b) No. The PSAR will be updated to remove references to the CRDRL.

(c) Since the BWR-2 design was introduced.

2. What solutions were offered by G.E. to Applicant, other than simply ending the return?

Response: None.

3. Is there leakage of cold water through the CRD return line during normal reactor operation of a BWR/6 238" core BWR system when the return li... is provided?

Response: There are no BWR-6's with CRD return lines.

4. On page 2 of an attachment (7906130309 to a letter (790610308) from Sherwood (G.E.) to V. Stello (NRC) dated: 5/22/80, Titled, "Changed to the Return Line to the Reactor Vessel CRD Hydraulic Control System," it states the CRD movement "...may be somewhat slower since the exhaust header pressure is higher" where the CRD return line is removed.

(a) What is the minimum and maximum the CRD movement will be slowed?

(b) What is meant by "stable without the reutnr line" in the same paragraph on Page 2?

(c) What is the basis for the summary statement on P. 5, "CRD performance without return line flow is satisfactory"?

Response:

(a) The referenced sentence refers to rod notch speeds. Subsequent confirmatory testing at operating plants showed no reduction of rod notch speeds.

(b) The purpose of the CRD return line was to accomodate flow/pressure fluctuations in the CRD system. Confirmatory tests at operating plants have shown that the CRD system is stable without the pressure of the CRDRL.

(c) Extensive CRD system evaluation and operating plant confirmatory tests substantiate the conclusion that none of the CRD system modes of operation are degraded by the elimination of the CRDRL.

5. What is the longest time ACNGS will be permitted to operate with the HPCS out of service?

Response: The period of time in question will be stipulated in the as yet unwritten Technical Specifications for ACNGS.

Contention 47 - Turbine Missiles

1. Does Applicant believe the current level of failure by hurling of turbine blocks or fragments is so low that no modification need be done as to orientation or structure of the ACNGS power block?

Response: Yes.

2. Does Applicant tkae the position a turbine missile cannot penetrate the turbine building?

Response: No.

3. What is the probability that a turbine missile will penetrate the turbine case?

Response: The aggregate average annual probability of wheel failure (and hence the probability that the turbine missile will penetrate the turbine case) is 1.1×10^{-8} (Refer to PSAR Table 3.5-2).

4. Are spare turbine spindles available for replacement from Applicant's supplier or in theevent of defect, will a spindle have to be fabricated upon receipt of an order from Applicant?

Response: No. A replacement spindle would have to be reordered.

5. What is the metalurgical composition of the turbine discs of ACNGS?

Response: The turbine discs are made of 12CR alloy steel.

Baker Contention 1 - Applicant financial qualifications

1. As pointed out in Baker 1 (Sept. 18, 1979), the Applicant's chief financial officer has taken the position before the PUC that "100% inclusion (of CWIP in the rate base) is required to...enable the Company to achieve its financial integrity requirements." The rate increase requested by Applicant in 1979 is still being appealed before the PUC.

(a) Does Applicant agree that a 50% level of CWIP in the rate base, when the allowed return on common equity, and level of operating expenses are identical to 1979, and the magnitude of CWIP expenditures allowed in the rate base equals 75% of the CWIP allowed in the rate base, would constitute a severe threat to its financial integrity?

(b) If Applicant does not "agree" with "(a)" <u>supra</u>., does Applicant <u>believe</u> that the conditions stated in "(a)" would constitute a severe threat to its financial integrity?

(c) Does Applicant agree that a 50% level of CWIP in the rate base, when the allowed return on common equity, and level of operating expenses are identical to 1979, and the magnitude of CWIP expenditures allowed in the rate base equals 100% of the CWIP allowed in the rate base, would constitute a severe threat to its financial integrity?

(d) If Applicant does not "agree" with "(c)" <u>supra</u>., does Applicant <u>believe</u> that the conditions stated in "(c)" would constitute a severe threat to its financial integrity?

(e) Does Applicant agree that a 50% level of CWIP in the rate base, when the allowed return on common equity, and level of operating expenses are identical to 1979, and the magnitude of CWIP expenditures allowed in the rate base equals 50% of the CWIP allowed in the rate base, would constitute a severe threat to its financial integrity?

(f) If Applicant does not "agree" with "(e)" supra, does Applicant believe that the conditions stated in "(e)" would constitute a severe threat to its financial integrity?

Response: Applicant cannot answer these interrogatories because they are internally inconsistent with, consequently, incomprehensible. For each of the scenarios listed Intervenor has made two different simultaneous assumptions about the amount of CWIP allowed in the rate base; this inconsistency makes the inquiry unanswerable at the start. Moreover, a significant number of other items besides the return on equity and level of operating expenses must be known before the level of CWIP allowed takes on any meaning. Intervenor has not supplied the necessary salient rate components which would make a conclusion or belief rationally based.

All documents referenced in these answers will be made available for inspection and copying at the Energy Development Complex. No presently identified expert witnesses answered any of these questions. Applicant recognizes its obligation to supplement or amend these answers in light of further "work" and will do so where and when appropriate.

Respectfully submitted,

OF COUNSEL:

BAKER & BOTTS
3000 One Shell Plaza
Houston, Texas 77002

Lowenstein, Newman, Reis, Axelrad & Toll
1025 Connecticut Ave., N.W.
Washington, D.C. 20036

C. Thomas Biddle, Jr.

J. Gregory Copeland
C. Thomas Biddle, Jr.
Darrell Ha[illegible]ck
3000 One Shell Plaza
Houston, Texas 77002

Jack R. Newman
Robert H. Culp
David Raskin
1025 Connecticut Ave., N.W.
Washington, D.C.

ATTORNEYS FOR
HOUSTON LIGHTING & POWER COMPANY

STATE OF TEXAS §
§
COUNTY OF HARRIS §

BEFORE ME, THE UNDERSIGNED AUTHORITY, on this day personally appeared Thomas E. Braudt, who upon his oath stated that he has answered the foregoing Houston Lighting & Power Company's Response to Doherty's Fourteenth Set of Interrogatories to Houston Lighting & Power Company in his capacity as Project Licensing Engineer for Houston Lighting & Power Company, and all statements contained therein are true and correct to the best of his knowledge and belief.

Thomas E. Braudt

Thomas E. Braudt

SUBSCRIBED AND SWORN TO BEFORE ME by the said Thomas E. Braudt, on this 24th day of June, 1980.

Cheryl A Southworth

Notary Public in and for
Harris County, Texas

Cheryl A Southworth

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	§	
	§	
HOUSTON LIGHTING & POWER COMPANY	§	Docket No. 50-466
	§	
(Allens Creek Nuclear Generating Station, Unit No. 1)	§	

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing Applicant's Response to John F. Doherty's Fourteenth Set of Interrogatories to Houston Lighting & Power Company in the above-captioned proceeding were served on the following by deposit in the United States mail, postage prepaid, or by hand-delivery this 24th day of June, 1980.

Sheldon J. Wolfe, Esq., Chairman
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. E. Leonard Cheatum
Route 3, Box 350A
Watkinsville, Georgia 30677

Mr. Gustave A. Linenberger
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. Chase R. Stephens
Docketing and Service Section
Office of the Secretary of the
Commission
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Steve Sohinki, Esq.
Staff Counsel
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Richard Lowerre, Esq.
Assistant Attorney General
for the State of Texas
P. O. Box 12548
Capitol Station
Austin, Texas 78711

Hon. Charles J. Dusek
Mayor, City of Wallis
P. O. Box 312
Wallis, Texas 77485

Hon. Leroy H. Grebe
County Judge, Austin County
P. O. Box 99
Bellville, Texas 77418

Atomic Safety and Licensing
Appeal Board
U.S. Nuclear Regulatory
Commission
Washington, D. C. 20555

Carro Hinderstein
609 Fannin, Suite 521
Houston, Texas 77002

Mr. Bryan L. Baker
1118 Montrose
Houston, Texas 77019

Stephen A. Doggett, Esq.
P. O. Box 592
Rosenberg, Texas 77471

Robert S. Framson
Madeline Bass Framson
4822 Waynesboro
Houston, Texas 77035

Mr. W. Matthew Perrenod
4070 Merrick
Houston, Texas 77025

Mr. James M. Scott
13935 Ivy Mount
Sugar Land, Texas 77478

D. Marrack
420 Mulberry Lane
Bellaire, Texas 77401

Mr. J. Morgan Bishop
11418 Oak Spring
Houston, Texas 77043

Mr. John F. Doherty
4327 Alconbury
Houston, Texas 77021

Ms. Brenda McCorkle
6140 Darnell
Houston, Texas 77074

Mr. Wayne E. Rentfro
P. O. Box 1335
Rosenberg, Texas 77471

Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory
Commission
Washington, D.C. 20555

C. Thomas Biddle Jr.

C. Thomas Biddle, Jr.